National Nuclear Security Administration
Department of Energy
National Nuclear Security Administration
Washington, DC 20585

JUN 2 6 2012

The Honorable Peter S. Winokur
Chairman
Defense Nuclear Facilities Safety Board
625 Indiana Avenue, NW, Suite 700
Washington, DC 20004

Dear Mr. Chairman:

This letter is in response to your letter of February 28, 2012, and staff report which documented concerns with the safety analysis for the Annular Core Research Reactor at Sandia National Laboratories (SNL). I share your concerns on the importance of an adequate safety analysis to ensure an appropriate control set is identified and implemented for these operations.

Enclosed is a consolidated response from the Sandia Site Office (SSO) and SNL providing current and planned actions to address the Board's concerns as well as to follow up on discussions from our April meeting. The SSO will continue to work with and coordinate discussions with your staff and SNL to ensure adequate resolution of these concerns.

If you have any questions on this matter, please contact Mr. James McConnell at (202) 586-4379.

Sincerely,

DONALD L. COOK
Deputy Administrator
for Defense Programs

Enclosures

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Sandia Site Office Report
To the
Defense Nuclear Facilities Safety Board
In Response to the
Letter Regarding Concerns with the
Annular Core Research Reactor,
Technical Area V,
Sandia National Laboratories

Sandia Site Office
Albuquerque, New Mexico

May 2012

Summary of Changes

Revision No.	Revision Description	Affected Pages
Rev. Final	Original Issue	All
Rev. 1	Incorporates comments from numerous sources	All

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Introduction

This report responds to the Defense Nuclear Facilities Safety Board (DNFSB) letter of February 28, 2012, regarding concerns with the Annular Core Research Reactor (ACRR) at Sandia National Laboratories (Sandia), Technical Area V (TA-V). Established in 1967, with the current core operational in 1978, the ACRR is a pool – type research reactor and Hazard Category 2 nuclear facility that is utilized for radiation effects testing and qualification of electronic systems and components. The ACRR is managed by Sandia Corporation (Sandia), Radiation Sciences Center that reports directly to Sandia's Science and Technology and Research Foundations Division.

In 2011, DNFSB Staff (Staff) personnel reviewed documents and completed two on-site visits with Sandia's and Sandia Site Office (SSO) technical staff during the weeks of July 25, 2011, and November 14, 2011. The activities were related to the ACRR safety basis, Instrumentation and Control, and Quality Assurance/Software Quality Assurance. This resulted in a comprehensive review of the last major revision of the ACRR Documented Safety Analysis (DSA), which was approved by the SSO in May 2007. SSO and Sandia maintained an open dialog with the Staff during the review, with a focus on continuous improvement. The Staff shared a number of issues and concerns as part of their out-brief, and formally provided these in the letter of February 28, 2012. As a direct result of ongoing improvement efforts and discussions with SSO, Sandia declared two Potentially Inadequate Safety Analyses (PISAs).

In addition to this report, the February 28, 2012 letter requested a briefing to discuss the National Nuclear Security Administration's (NNSA) path forward. The briefing was conducted with the Board on April 26, 2012 at DNFSB Headquarters in Washington, DC. At that time, additional clarification was requested on certain issues, and is addressed in this report.

The SSO and Sandia recognize the value of the review conducted by the DNFSB Staff. While SSO and Sandia do not agree with every statement in the DNFSB letter and Staff Issue Report, we do acknowledge that there are a number of opportunities for improvement that can be integrated with the improvement strategy at the ACRR. Similarly, Sandia began drafting their Improvement Plan after the conclusion of the second site visit in November 2011 to address some of the issues and concerns identified by the Staff. A video teleconference was held in December 2011 to seek clarity on the issues, and Sandia provided a written synopsis to the Staff to ensure Sandia accurately captured the Staff's issues. Sandia finalized revision 0 of the formal Improvement Plan on April 24, 2012.

The letter and Staff Issue Report also referenced the potential for additional correspondence addressing quality assurance and software quality assurance. The DNFSB letter on those subjects was issued on April 18, 2012. Sandia's Improvement Plan includes actions addressing some of the concerns with the Quality Assurance and Software Quality Assurance programs, but the response to the April 18, 2012 letter will be addressed in a separate briefing and correspondence.

Issues Related to the Documented Safety Analysis

The Staff were concerned that the current accident analyses for the ACRR did not use reasonably conservative or bounding quantities of material at risk (MAR), accident analyses, or release parameters. The issues identified in the Board Staff Issue Report are paraphrased below, followed by a discussion of the facts.

1. Issue Statement: The most severe of the Design Basis Accidents (DBAs) for reactor operations assumes significant reactor fuel melting; however, the consequence analysis does not account for the presence of experimental MAR in the central cavity during these DBAs. Similarly, the Documented Safety Analysis (DSA) does not include a calculation of the amount of material, either fuel or MAR, that can be vaporized under DBA conditions.

Discussion: SSO agrees with the Staff that the ACRR DSA did not analyze the conditions described and this is an inherent feature of the safe harbor methodology for operational design basis accidents. Regulatory Guide (RG) 1.70 is the specified safe harbor for a DOE reactor which identifies the specific type of accidents to be analyzed as they pertain to reactor operations, but does not lend itself to accidents associated with the experiments. DOE-STD-3009 was used to analyze the accidents associated with the experiments and based on that methodology the experiment events are considered unmitigated without consideration for how the reactor is performing.

Concerning the vaporization of Pu, the amount was intentionally limited based on historical fielded experiments; not the maximum amount of bare Pu metal that could hypothetically be vaporized. Further, the DSA identifies energy deposition limits that allow for the melting of fissionable material, but, precludes Pu vaporization.

Sandia has acknowledged the concern regarding the combined effect of the reactor event with the MAR, declared a PISA, implemented compensatory measures related to this concern, and has developed a path forward to reevaluate the combined effect of accident conditions at the ACRR.

Additionally, as part of their Improvement Plan, Sandia will re-visit specific events to determine under what conditions (power generation & energy deposition) and in what type of experiments (configuration & amount of MAR) could vaporization occur.

2. Issue Statement: The safety analysis relies on a computer code to determine the extent of fuel melting during accidents. Fuel melting would lead to significant water boiling at the surface of the fuel rods. Sandia analysts failed to validate the code in that regime (where fuel melts under accident conditions).

Discussion: Sandia performed an evaluation of a hypothetical "Loss of Pool Water" accident in the ACRR (2005), which concluded that, although some fuel elements could be damaged, no cladding would be breached (no fuel melting). Although some accident scenarios in the DSA have not analyzed the percentage of fuel expected to fail, the quantity of fuel failing is used or postulated as an entry condition, and in order to facilitate calculations to determine expected releases and off-site consequences. Operating limits and safety system setpoints have been selected well below predicted failure points to ensure sufficient margin to fuel failure in postulated temperature transients. Accident analysis codes have been validated in the normal operating regimes. The large margin between predicted fuel failure and operating limits compensate for the uncertainty introduced by the extrapolation of the code results from normal operating conditions to accident conditions.

As part of the Improvement Plan, Sandia will upgrade the reactor kinetics/thermodynamic code to meet the properly graded Safety Software and Quality Assurance standards and reanalyze high-priority accidents. The code will be validated against actual ACRR performance data in the nominal pulse and steady-state modes.

3. Issue Statement: The DSA does not contain limits on the amount of fuel in the storage pool, other than the geometric constraints of the racks in the pool. While the pool currently contains no fuel, the DSA authorizes such storage. None of the DBAs include the consequences from insults to the fuel in the storage pool.

Discussion: SSO agrees with the Staff that there are no TSR limits on the storage pool, as established by the hazards and accident analyses. Less than 2 percent of the core has been used (burned up) since operations began in 1978. The ACRR DSA accident analysis assumptions bound the maximum fission product inventory at 15 megacuries (i.e., based on a continuous five year maximum power history to create this inventory). The ACRR fuel does not present the same types of concerns that other reactor fuels create (e.g. decay heat, large fission product inventories, etc.). Movement of fuel from the reactor core to the storage pool using a fuel element transfer rack has been analyzed. The following events have been analyzed in the DSA and are bounded for the storage of fuel in the pool:

- CF-INT-003 "Crane drop/impact causes a transfer rack impact during movement of new or spent reactor fuel in the high bay, resulting in the potential release of radioactive materials/contamination,"
- CF-INT-007 "A heavy load is dropped onto the FREC-II or ACRR core or stored fuel elements in the ACRR pool or storage pool resulting in the potential release of radioactive materials/contamination", and
- RP-INT-002 "A loss of water in the ACRR pool and storage pool caused the reactor core to become exposed, resulting in a facility worker exposure to an unshielded radiation field."

4. Issue Statement: The discussion of the beyond design basis accident (BDBA), a seismic event with complete loss of reactor pool water, concludes that no damage or release would occur as a result of the accident. The Staff does not believe that the postulated BDBA scenario is appropriate for the facility. Operational BDBAs are those operational accidents with more severe conditions or equipment failures than are estimated for the corresponding DBA. The BDBA currently presented in the DSA is not consistent with either RG 1.70 or DOE-STD-3009. Therefore, the BDBA does not provide insight into the identification of facility features that could provide additional prevention or mitigation of accidents with severe consequences.

Discussion: SSO recognizes the concept of analyzing the BDBA to reinforce the determination of facility features to provide additional prevention or mitigation. The design of the ACRR fuel for pulse operations to high temperatures makes it very robust to accident conditions. The current BDBA postulates a very unlikely scenario in which a seismic event occurs that is significant enough to cause a leak in the reactor pool, however the event does not result in mechanical disruption of the core (i.e., core remains intact). These two conditions would be unlikely to occur from the same event; however, the BDBA assumes the ACRR remains in a critical state while the pool drains. Due to the robust design of the fuel and the nature of the reactor to lose power and shutdown as the pool water level decreases, fuel cladding integrity is maintained under these conditions. In contrast, the seismic design basis event in the DSA uses mechanical disruption of the core resulting in fuel cladding damage as an entry condition to the scenario, thus a release of fission products is postulated for this event.

SSO will be coordinating with NNSA on the strategy to address methods for analyzing BDBAs. The strategy will be provided to Sandia so that the BDBA selection and analysis can be re-visited.

5. Issue Statement: The selection of specific parameters for consequence analysis is judged to be non-conservative. Two references used in the DSA recommend different pool release fractions. Sandia analysts have chosen to use the less conservative pool release fractions given by Powers while offering a limited technical basis for that decision. Similarly, the Staff noted that Sandia had not provided an adequate justification for using the less-conservative dry deposition velocity. Lastly, regarding airborne release fraction and respirable fraction, Sandia calculated the consequences to the public due to the experimental MAR in the facility fire accident analysis (self- sustained oxidation of Pu metal), and applied the same consequences to the aircraft crash and earthquake accident scenarios. The analysis did not account for mechanical dispersion and blast effects. And, an administrative control allows 1 g of Pu-239 to be stored contiguous with explosives in the facility. The consequence analysis does not account for this plutonium, which would yield a significantly higher airborne release fraction and respirable fraction in an explosion than it would under self-sustained oxidation.

Discussion: SSO agrees with the Staff that refinement of the analysis will bring clarity to this issue. After reviewing the details of the issue, Sandia declared a PISA related to the MAR and explosives. The safety evaluation resulting from the PISA determined a minor increase in off-site consequences for associated accidents; however the overall offsite dose is still less than one rem. This increase did not result in the need for any additional controls, or challenge the Evaluation Guideline. However, related to Pool Release Fractions, SSO's opinion is that the choice of values for the Pool Release Fraction was appropriate. The value used was selected from a study (Powers) referenced by the Staff that is referenced in the ACRR DSA and was specific to ACRR accident conditions. This value becomes even more conservative when it is realized that no plate-out is credited, as in other analyses. SSO's opinion is that Sandia performed their due diligence.

Regarding the dry deposition velocity, Sandia did complete a "New Information" form in August 2011 to evaluate the HSS Safety Bulletin. In addition, Sandia already uses the more conservative 0.0 m/s deposition velocity value for Tritium and gases. Sandia, along with SSO and other NNSA sites are supporting NNSA headquarters in evaluating HSS-2011-02 and formulating a path forward for proper implementation, as necessary.

The ACRR experimental source term for aircraft crash and earthquake accidents is actinide metals, and Pu metal is selected to be representative of experimental materials. For these accident scenarios, the initial insult to the material is postulated to be mechanical impact due to the ACRR structure collapsing on the Pu or direct impact from aircraft debris. Per DOE-HDBK-3010-94, mechanical impact on metals results in a negligible ARF/RF. Both the aircraft crash and earthquake scenario postulate a secondary facility fire. The DSA conservatively assumes the entire facility inventory is impacted by the secondary fire. Per DOE-HDBK-3010-94 for thermal impact on metal (self-sustained oxidation) the ARF/RF is 2.5×10^{-4}. The ACRR DSA uses this ARF/RF for the earthquake and aircraft crash scenarios.

The ACRR DSA authorizes the storage of up to 500 g of TNT equivalent. However, the DSA limits the amount of material that can be contiguous with the explosives to 1 g of Pu-239 equivalent. During an earthquake induced fire or an aircraft crash, the 500 g of explosive may detonate. Per DOE-HDBK-3010-94, the explosive impact on metal results in a negligible ARF/RF.

The aircraft crash and earthquake scenarios are reasonably conservative in their assumptions and postulated release mechanisms. The scenarios select the bounding ARF/RF values from DOE-HNBK-3010-94 for the given form of material and impacts.

As part of the Improvement plan, Sandia will update analytical tools to develop a coupled reactor transient/fissile experiment model. The model will be used to conservatively predict reactor experiment impacts for credible reactor transients. The results of the analysis will be incorporated into the accident analysis of the DSA as appropriate.

Issues Related to the Adequacy of Reactor Controls

The Staff identified concerns with the application of design standards for the Instrumentation and Control system, and issues of reliability in the Reactivity Control System (RCS). The issues identified in the Board Staff Issue Report are paraphrased below, followed by a discussion of the facts.

1. Issue Statement: The DSA for the ACRR references American National Standards Institute (ANSI)/American Nuclear Society (ANS) 15.15, *Criteria for the Reactor Safety Systems of Research Reactors, in relation to the design of the Plant Protect System (PPS).* This standard specifies that the PPS must meet single-failure criteria and establishes additional independence requirements. The PPS, however, does not meet single-failure criteria. Sandia personnel cited the exception in ANSI/ANS-15.15 that compliance with single-failure criteria is not mandatory for research reactors posing negligible risk. The Staff concludes, however, that ACRR operations pose a non-negligible risk to workers and the public. The DSA for the ACRR states that the control panel indications used to alert operators to the presence of a fault mitigate the significance of not meeting the single-failure criteria and independence requirements of

ANSI/ANS-15.15 for the PPS. However, it does not define specific operator actions required in response to abnormal indications as part of a credited safety function.

Discussion: SSO agrees with the Staff that Sandia did not clearly identify in the DSA, which portions of ANS 15.15 are applicable. As pointed out during the review, the standard was not adopted in its entirety. Sandia references ANS 15.15 to emphasize that a significant amount of rigor was placed into the design of the PPS. Those portions of ANS 15.15 that were applied, along with the justification for that selection, will be documented in the System Design Description as part of the Improvement Plan.

Sandia uses the criteria in DOE-STD-3009 to determine the categorization of equipment as Safety Class or Safety Significant. ACRR has no "Safety Class" equipment, since none of the postulated accidents challenge the Evaluation Guideline. The credited safety systems are "Safety Significant" and, as such, are not required to meet the redundancy requirements of "Safety Class" equipment which is analogous to single-point failure criteria.

Additionally, regarding the Staff's comments about specific operator actions, the DSA does not credit operator actions for unmitigated accidents. None of the ACRR's unmitigated accidents challenge the Evaluation Guideline. The failure of the mode selector switch for the Plant Protection System is recognized as a single-point failure. However, any operator actions taken to prevent/mitigate an accident are not credited in the DSA. SSO's opinion is it would be inappropriate to define operator actions in response to abnormal indications that are not credited to prevent/mitigate an accident in the DSA. This level of detail is captured by operating level procedures.

2. Issue Statement: Since the Reactor Console/Rod Control Upgrade was completed in 2002, several problems with components within the Reactivity Control System (RCS) have arisen. Some of these problems resulted in a simple system lockup (at least five instances) with little safety impact. Others resulted in uncontrolled rod motion (at least two instances), effectively, but briefly, initiating the design basis rod withdrawal accident scenario. Based on the observed component failure rates during the last 10 years, the staff does not consider the RCS to be sufficiently reliable to perform its safety-significant function. The Staff notes the poor performance of the RCS presents compelling evidence of the need to consider a more formal analysis of system reliability and operability.

Discussion: SSO shares the Staff's concerns related to equipment failures and anomalies, but does not share their opinion related to the equipment performing its safety function. The hazard analysis does identify scenarios that evaluate failures associated with the regulating and FREC-II rods (DBA involves entire regulating rod bank), regardless of the initiating event. The robustness of the reactor core and fuel element designs are inherent conditions considered in the analysis that concludes no radiological release occurs. Furthermore, unlike commercial nuclear power reactors, the small ACRR research reactor does not generate sustained large decay heat buildup, which further reduces the risk of a release from inadvertent rod movement. The events were carried forward to the accident analysis to meet the requirements of both RG 1.70 and NUREG-1537.

SSO believes the RCS system has demonstrated satisfactory performance when the numerous startups/shutdowns conducted each operational day are considered. For example, each Pulse and Steady State operation requires the minimum movement of the Safety, Control and Transient Rods as shown in Table 1 below.

Table 1: Regulating Rod Movements for ACRRF

Regulating Rod (RR) (# of each)	Movement during a single PULSE / Movement during Steady State Operation	Number of RR Motions Pulse / Steady-State
Safety Rods (2)	*Fully Withdrawn once *Fully Withdrawn once	2 RR motions 2 RR motions
Transient Rods (3)	*Three separate positioning for UP and DOWN Delayed Critical measurements, and final setup position *Fully Withdrawn once	9 RR motions 3 RR motions
Control Rods (6)	*Three separate positioning for UP and DOWN Delayed Critical measurements, and final setup position *For conservatism, stating 1 withdraw to desired power level	18 RR motions 6 RR motions

Therefore, a minimum of 29 regulating rod motions must be successfully completed for each Pulse operation and a minimum of 11 regulating rod motions for each Steady State Operation. This summary only considers large rod motions and does not include the numerous shim command applied during operations to adjust reactor power and start up rate. These numbers do not include conduct of ACRR-OP-001 Pre Operation Checkout in which regulating rods are tested each operational day.

In CY2010, 247 Pulse and 89 Steady State Operations were conducted resulting in (247×29) 7,163 Pulse and (89×11) 979 Steady State regulating rod motions, for a total of 8,142 regulating rod motions. Seven regulating rod anomalies were noted in the Pulse and Steady State procedures in CY2010 resulting in a failure rate of (7/8142) 0.086 percent.

The ACRR Plant Protection System (PPS) provides the credited, safety-significant prevention feature to ensure the safe operation of the reactor. The PPS is independent of, and isolated from, the RCS. The PPS uses analog inputs from fuel thermocouples, dedicated fission monitors and open analog relays to de-energize the magnet power supply if setpoints are exceeded. The reliability of the RCS does present an impact to mission-related operations, and as part of the Improvement Plan, Sandia plans to better distinguish between safety and operations control functions and track the system performance based on the safety function to address the issue of

RCS reliability. A system reliability analysis will be completed and formal reliability goals established for the systems. The resolution of the design standard concern will involve providing clear bases for the graded approach of the standards.

SSO acknowledges improvements in the System Design Descriptions (SDDs) are necessary to clearly define those elements of the RCS design that support a credited safety function. The current SDD was developed after the design and installation of the RCS. Sandia has committed to strengthen the link between the design standards, safety functions, and performance criteria for the RCS system.

Issues Related to the Adequacy of Safe Harbor Methodology

The Staff was concerned that the safe harbor methodology used for the ACRRF safety basis may not be appropriate. The issues identified in the Board Staff Issue Report are paraphrased below, followed by a discussion of the facts.

1. Issue Statement: During this review, the Board's staff determined that NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors* (1996), may be a more appropriate safe harbor for test reactors such as the ACRR. NRC regulators use NUREG-1537 for the licensing of new non-power reactors. Although the Nuclear Safety Management Rule (10 CFR 830) provides the option of using a "successor document" to Regulatory Guide 1.70, the contractor did not exercise this option. Also, the Staff noted that several of the issues related to the DSA for the ACRR could have been avoided if NUREG-1537 had been consulted at the time the DSA was developed.

Discussion: SSO acknowledges the Staff's comments considering NUREG 1537 as an alternative methodology as allowed for using the 10 CFR 830 exemption process. However, as part of the DSA, Sandia did take into consideration the application of other guides and standards as noted in the Accident Selection section of the ACRR DSA, which states;

> In addition to the unique and bounding accidents, a review was conducted of guides and standards applicable to research reactor safety analysis, including; ANSI/ANS 15.21-1996, *Format and Content for Safety Analysis Reports for Research Reactors*, Regulatory Guide 1.70, Revision 3, *Standard Format and Content of the Safety Analysis Reports for Nuclear Power Plants*, LWR Edition, and NUREG-1537 Part 1, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*, Format and Content. Sandia conducted this review to ensure the accident analysis selection was thorough and complete.

The ACRR DSA also documents this consideration of the application of other guides and standards as noted in Section 16.6.6.5 of the ACRR DSA, which states;

> In NRC licensing schemes, as described by NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,* LCOs are required on coolant temperature, level, and water chemistry. In the spirit of the NRC requirements, the ACRRF has chosen to apply TSR controls on these parameters.

Although the Staff implies otherwise, it should be noted that NUREG-1537 is not a successor document to RG 1.70 and does not address or analyze the use of experiments as a part of the reactor operations. Therefore, the ACRR would still require DOE-STD-3009 as an additional Safe Harbor. Another important fact is that when 10 CFR 830 was implemented in 2003, the process allowed for the selection of "alternate methodologies" in lieu of the Safe Harbors in the Rule, but the vast majority of sites did not pursue this path because early guidance indicated much scrutiny would be applied, and few would be approved.

As identified in the Improvement Plan, Sandia has committed to re-evaluate the applicability of NUREG 1537 as part of an overall effort to improve the DSA.

Conclusion

SSO and Sandia recognize the enduring ACRR mission into the indefinite future. SSO and Sandia consider ACRR operations to be safe, and are committed to continuous, ongoing evaluations to ensure a safe operating envelope is maintained, all requirements are being met, and that the outcome provides a robust and accurate safety basis. The results of the Staff review will be utilized by SSO and Sandia, in conjunction with ongoing efforts, to identify areas in the ACRR DSA that can benefit from a re-evaluation and update. SSO and Sandia continue to maintain and further develop the Institute of Nuclear Power Operations Principles for a Strong Nuclear Safety Culture and welcome this opportunity to improve.